total plant equipment that is out of service should be taken into account to determine the overall effect on performance of safety functions.

(b) The scope of the monitoring program specified in paragraph (a)(1) of this section shall include safety related and nonsafety related structures, systems, and components, as follows:

(1) Safety-related structures, systems and components that are relied upon to remain functional during and following design basis events to ensure the integrity of the reactor coolant pressure boundary, the capability to shut down the reactor and maintain it in a safe shutdown condition, or the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to the guidelines in § 50.34(a)(1) or § 100.11 of this chapter, as applicable.

(2) Nonsafety related structures, systems, or components:

(i) That are relied upon to mitigate accidents or transients or are used in plant emergency operating procedures (EOPs); or

(ii) Whose failure could prevent safety-related structures, systems, and components from fulfilling their safety-related function; or

(iii) Whose failure could cause a reactor scram or actuation of a safety-related system.

(c) The requirements of this section shall be implemented by each licensee no later than July 10, 1996.

[56 FR 31324, July 10, 1991, as amended at 58 FR 33996, June 23, 1993; 61 FR 39301, July 29, 1996; 61 FR 65173, Dec. 11, 1996; 62 FR 47271, Sept. 8, 1997; 62 FR 59276, Nov. 3, 1997]

§ 50.66 Requirements for thermal annealing of the reactor pressure vessel.

(a) For those light water nuclear power reactors where neutron radiation has reduced the fracture toughness of the reactor vessel materials, a thermal annealing may be applied to the reactor vessel to recover the fracture toughness of the material. The use of a thermal annealing treatment is subject to the requirements in this section. A report describing the licensee's plan for conducting the thermal annealing must be submitted in accordance with § 50.4 at least three years prior to the date at which the limiting fracture toughness criteria in § 50.61 or appendix G to part 50 would be exceeded. Within three years of the submittal of the Thermal Annealing Report and at least thirty days prior to the start of the thermal annealing, the NRC will review the Thermal Annealing Report and place the results of its evaluation in its Public Document Room. The licensee may begin the thermal anneal after:

(1) Submitting the Thermal Annealing Report required by paragraph (b) of this section;

(2) The NRC places the results of its evaluation of the Thermal Annealing Report in the Public Document Room; and

(3) The requirements of paragraph (f)(1) of this section have been satisfied.

(b) *Thermal Annealing Report.* The Thermal Annealing Report must include: a Thermal Annealing Operating Plan; a Requalification Inspection and Test Program; a Fracture Toughness Recovery and Reembrittlement Trend Assurance Program; and Identification of Unreviewed Safety Questions and Technical Specification Changes.

(1) Thermal Annealing Operating Plan.

The thermal annealing operating plan must include:

(i) A detailed description of the pressure vessel and all structures and components that are expected to experience significant thermal or stress effects during the thermal annealing operation;

(ii) An evaluation of the effects of mechanical and thermal stresses and temperatures on the vessel, containment, biological shield, attached piping and appurtenances, and adjacent equipment and components to demonstrate that operability of the reactor will not be detrimentally affected. This evaluation must include:

(A) Detailed thermal and structural analyses to establish the time and temperature profile of the annealing operation. These analyses must include heatup and cooldown rates, and must demonstrate that localized temperatures, thermal stress gradients, and subsequent residual stresses will not result in unacceptable dimensional changes or distortions in the vessel, attached piping and appurtenances, and that the thermal annealing cycle will not result in unacceptable degradation of the fatigue life of these components.

(B) The effects of localized high temperatures on degradation of the concrete adjacent to the vessel and changes in thermal and mechanical properties, if any, of the reactor vessel insulation, and on detrimental effects, if any, on containment and the biological shield. If the design temperature limitations for the adjacent concrete structure are to be exceeded during the thermal annealing operation, an acceptable maximum temperature for the concrete must be established for the annealing operation using appropriate test data.

(iii) The methods, including heat source, instrumentation and procedures proposed for performing the thermal annealing. This shall include any special precautions necessary to minimize occupational exposure, in accordance with the As Low As Reasonably Achievable (ALARA) principle and the provisions of § 20.1206.

(iv) The proposed thermal annealing operating parameters, including bounding conditions for temperatures and times, and heatup and cooldown schedules.

(A) The thermal annealing time and temperature parameters selected must be based on projecting sufficient recovery of fracture toughness, using the procedures of paragraph (e) of this section, to satisfy the requirements of § 50.60 and § 50.61 for the proposed period of operation addressed in the application.

(B) The time and temperature parameters evaluated as part of the thermal annealing operating plan, and supported by the evaluation results of paragraph (b)(1)(ii) of this section, represent the bounding times and temperatures for the thermal annealing operation. If these bounding conditions for times and temperatures are violated during the thermal annealing operation, then the annealing operation is considered not in accordance with the Thermal Annealing Operating Plan, as required by paragraph (c)(1) of this section, and the licensee must comply with paragraph (c)(2) of this section.

(2) Requalification Inspection and Test Program. The inspection and test program to requalify the annealed reactor vessel must include the detailed monitoring, inspections, and tests proposed to demonstrate that the limitations on temperatures, times and temperature profiles, and stresses evaluated for the proposed thermal annealing conditions of paragraph (b)(1)(iv) of this section have not been exceeded, and to determine the thermal annealing time and temperature to be used in quantifying the fracture toughness recovery. The requalification inspection and test program must demonstrate that the thermal annealing operation has not degraded the reactor vessel, attached piping or appurtenances, or the adjacent concrete structures to a degree that could affect the safe operation of the reactor.

(3) Fracture Toughness Recovery and Reembrittlement Trend Assurance Program. The percent recovery of RT_{NDT} and Charpy upper-shelf energy due to the thermal annealing treatment must be determined based on the time and temperature of the actual vessel thermal anneal. The recovery of RT_{NDT} and Charpy upper-shelf energy provide the basis for establishing the post-anneal RT_{NDT} and Charpy upper-shelf energy for each vessel material. Changes in the RT_{NDT} and Charpy upper-shelf energy with subsequent plant operation must be determined using the post-anneal values of these parameters in conjunction with the projected reembrittlement trend determined in accordance with paragraph (b)(3)(ii) of this section. Recovery and reembrittlement evaluations shall include:

(i) Recovery Evaluations. (A) The percent recovery of both RT_{NDT} and Charpy upper-shelf energy must be determined by one of the procedures described in paragraph (e) of this section, using the proposed lower bound thermal annealing time and temperature conditions described in the operating plan.

(B) If the percent recovery is determined from testing surveillance specimens or from testing materials removed from the reactor vessel, then it shall be demonstrated that the proposed thermal annealing parameters used in the test program are equal to or bounded by those used in the vessel annealing operation.

(C) If generic computational methods are used, appropriate justification must be submitted as a part of the application.

(ii) Reembrittlement Evaluations. (A) The projected post-anneal reembrittlement of RT_{NDT} must be calculated using the procedures in § 50.61(c), or must be determined using the same basis as that used for the pre-anneal operating period. The projected change due to post-anneal reembrittlement for Charpy upper-shelf energy must be determined using the same basis as that used for the pre-anneal operating period.

(B) The post-anneal reembrittlement trend of both RT_{NDT} and Charpy upper-shelf energy must be estimated, and must be monitored using a surveillance program defined in the Thermal Annealing Report and which conforms to the intent of Appendix H of this part, ''Reactor Vessel Material Surveillance Program Requirements.''

(4) Identification of Unreviewed Safety Questions and Technical Specification Changes. Any changes to the facility as described in the updated final safety analysis report constituting unreviewed safety questions, and any changes to the technical specifications, which are necessary to either conduct the thermal annealing or operate the nuclear power reactor following the annealing, must be identified. The section shall demonstrate that the Commission's requirements continue to be complied with, and that there is reasonable assurance of adequate protection to the public health and safety following the changes.

(c) *Completion or Termination of Thermal Annealing.* (1) If the thermal annealing was completed in accordance with the Thermal Annealing Operating Plan and the Requalification Inspection and Test Program, the licensee shall so confirm in writing to the Director, Office of Nuclear Reactor Regulation. The licensee may restart its reactor after the requirements of paragraph (f)(2) of this section have been met.

(2) If the thermal annealing was completed but the annealing was not performed in accordance with the Thermal Annealing Operating Plan and the Requalification Inspection and Test Program, the licensee shall submit a summary of lack of compliance with the Thermal Annealing Operating Plan and the Requalification Inspection and Test Program and a justification for subsequent operation to the Director, Office of Nuclear Reactor Regulation. Any changes to the facility as described in the updated final safety analysis report which are attributable to the noncompliances and constitute unreviewed safety questions, and any changes to the technical specifications which are required as a result of the noncompliances, shall also be identified.

(i) If no unreviewed safety questions or changes to technical specifications are identified, the licensee may restart its reactor after the requirements of paragraph (f)(2) of this section have been met.

(ii) If any unreviewed safety questions or changes to technical specifications are identified, the licensee may not restart its reactor until approval is obtained from the Director, Office of Nuclear Reactor Regulation and the requirements of paragraph (f)(2) of this section have been met.

(3) If the thermal annealing was terminated prior to completion, the licensee shall immediately notify the NRC of the premature termination of the thermal anneal.

(i) If the partial annealing was otherwise performed in accordance with the Thermal Annealing Operating Plan and relevant portions of the Requalification Inspection and Test Program, and the licensee does not elect to take credit for any recovery, the licensee need not submit the Thermal Annealing Results Report required by paragraph (d) of this section but instead shall confirm in writing to the Director, Office of Nuclear Reactor Regulation that the partial annealing was otherwise performed in accordance with the Thermal Annealing Operating

Plan and relevant portions of the Requalification Inspection and Test Program. The licensee may restart its reactor after the requirements of paragraph (f)(2) of this section have been met.

(ii) If the partial annealing was otherwise performed in accordance with the Thermal Annealing Operating Plan and relevant portions of the Requalification Inspection and Test Program, and the licensee elects to take full or partial credit for the partial annealing, the licensee shall confirm in writing to the Director, Office of Nuclear Reactor Regulation that the partial annealing was otherwise performed in compliance with the Thermal Annealing Operating Plan and relevant portions of the Requalification Inspection and Test Program. The licensee may restart its reactor after the requirements of paragraph (f)(2) of this section have been met.

(iii) If the partial annealing was not performed in accordance with the Thermal Annealing Operating Plan and relevant portions of the Requalification Inspection and Test Program, the licensee shall submit a summary of lack of compliance with the Thermal Annealing Operating Plan and the Requalification Inspection and Test Program and a justification for subsequent operation to the Director, Office of Nuclear Reactor Regulation. Any changes to the facility as described in the updated final safety analysis report which are attributable to the noncompliances and constitute unreviewed safety questions, and any changes to the technical specifications which are required as a result of the noncompliances, shall also be identified.

(A) If no unreviewed safety questions or changes to technical specifications are identified, the licensee may restart its reactor after the requirements of paragraph (f)(2) of this section have been met.

(B) If any unreviewed safety questions or changes to technical specifications are identified, the licensee may not restart its reactor until approval is obtained from the Director, Office of Nuclear Reactor Regulation and the requirements of paragraph (f)(2) of this section have been met.

(d) *Thermal Annealing Results Report.* Every licensee that either completes a thermal annealing, or that terminates an annealing but elects to take full or partial credit for the annealing, shall provide the following information within three months of completing the thermal anneal, unless an extension is authorized by the Director, Office of Nuclear Reactor Regulation:

(1) The time and temperature profiles of the actual thermal annealing;

(2) The post-anneal RT_{NDT} and Charpy upper-shelf energy values of the reactor vessel materials for use in subsequent reactor operation;

(3) The projected post-anneal reembrittlement trends for both RT_{NDT} and Charpy upper-shelf energy; and

(4) The projected values of RT_{PTS} and Charpy upper-shelf energy at the end of the proposed period of operation addressed in the Thermal Annealing Report.

(e) *Procedures for Determining the Recovery of Fracture Toughness.* The procedures of this paragraph must be used to determine the percent recovery of ΔRT_{NDT}, R_t, and percent recovery of Charpy upper-shelf energy, R_u. In all cases, R_t and R_u may not exceed 100.

(1) For those reactors with surveillance programs which have developed credible surveillance data as defined in § 50.61, percent recovery due to thermal annealing (R_t and R_u) must be evaluated by testing surveillance specimens that have been withdrawn from the surveillance program and that have been annealed under the same time and temperature conditions as those given the beltline material.

(2) Alternatively, the percent recovery due to thermal annealing (R_t and R_u) may be determined from the results of a verification test program employing materials removed from the beltline region of the reactor vessel[6] and that have been annealed under the same time and temperature conditions as those given the beltline material.

[6] For those cases where materials are removed from the beltline of the pressure vessel, the stress limits of the applicable portions of the ASME Code Section III must be satisfied, including consideration of fatigue and corrosion, regardless of the Code of record for the vessel design.

(3) Generic computational methods may be used to determine recovery if adequate justification is provided.

(f) *Public information and participation*. (1) Upon receipt of a Thermal Annealing Report, and a minimum of 30 days before the licensee starts thermal annealing, the Commission shall:

(i) Notify and solicit comments from local and State governments in the vicinity of the site where the thermal annealing will take place and any Indian Nation or other indigenous people that have treaty or statutory rights that could be affected by the thermal annealing,

(ii) Publish a notice of a public meeting in the FEDERAL REGISTER and in a forum, such as local newspapers, which is readily accessible to individuals in the vicinity of the site, to solicit comments from the public, and

(iii) Hold a public meeting on the licensee's Thermal Annealing Report.

(2) Within 15 days after the NRC's receipt of the licensee submissions required by paragraphs (c)(1), (c)(2) and (c)(3)(i)–(iii) of this section, the NRC staff shall place in the NRC Public Document Room a summary of its inspection of the licensee's thermal annealing, and the Commission shall hold a public meeting:

(i) For the licensee to explain to NRC and the public the results of the reactor pressure vessel annealing,

(ii) for the NRC to discuss its inspection of the reactor vessel annealing, and

(iii) for the NRC to receive public comments on the annealing.

(3) Within 45 days of NRC's receipt of the licensee submissions required by paragraphs (c)(1), (c)(2) and (c)(3) (i)–(iii) of this section, the NRC staff shall complete full documentation of its inspection of the licensee's annealing process and place this documentation in the NRC Public Document Room.

[60 FR 65472, Dec. 19, 1995]

§ 50.68 Criticality accident requirements.

(a) Each holder of a construction permit or operating license for a nuclear power reactor issued under this part, or a combined license for a nuclear power reactor issued under part 52 of this chapter shall comply with either 10 CFR 70.24 of this chapter or requirements in paragraph (b).

(b) Each licensee shall comply with the following requirements in lieu of maintaining a monitoring system capable of detecting a criticality as described in 10 CFR 70.24:

(1) Plant procedures may not permit handling and transportation at any one time of more fuel assemblies than have been determined to be safely subcritical under the most adverse moderation conditions feasible by unborated water.

(2) The estimated ratio of neutron production to neutron absorption and leakage (k-effective) of the fresh fuel in the fresh fuel storage racks shall be calculated assuming the racks are loaded with fuel of the maximum permissible U–235 enrichment and flooded with pure water and must not exceed 0.95, at a 95 percent probability, 95 percent confidence level.

(3) If optimum moderation of fresh fuel in the fresh fuel storage racks occurs when the racks are assumed to be loaded with fuel of the maximum permissible U–235 enrichment and filled with low-density hydrogenous fluid, the k-effective corresponding to this optimum moderation must not exceed 0.98, at a 95 percent probability, 95 percent confidence level.

(4) If no credit for soluble boron is taken, the k-effective of the spent fuel storage racks loaded with fuel of the maximum permissible U–235 enrichment must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with pure water. If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum permissible U–235 enrichment must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with pure water.

(5) The quantity of SNM, other than nuclear fuel stored on site, is less than the quantity necessary for a critical mass.

(6) Radiation monitors, as required by GDC 63, are provided in storage and associated handling areas when fuel is present to detect excessive radiation